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Abstract

GAS CIRCULATORS AND THEIR DRIVES FOR LARGE GAS-COOLED POWER REACTORS

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The two basic types of circulators and the many forms of drive available are investigated and discussed together with examples of their application and the reasons for the choice.

Centrifugal and axial circulators main features are described together with an extensive discussion on the relative merits of steam turbine drive, fixed speed a.c. electric, variable speed a.c. electric and variable speed d.c. motor drive.

Electric motor drive has the advantage of simplicity but constant speed a.c. motors demand either a gearbox or that the circulator speed be matched to a 2 or 4 pole motor which may compromise circulator design. Variable speed a.c. motors require either a variable frequency supply at high initial plant cost or brush gear of high current carrying capacity with attendant maintenance needs whilst d.c. motors have a limited capacity and require expensive rectification or conversion equipment.

Steam turbine drive proves attractive if the optimum level of h.p. steam generation matches the blower power when the exhaust pressure is made to coincide with the pressure of l.p. system in a dual pressure cycle. Reheating of the partly expanded h.p. steam and mixing with l.p. steam provides a neat and economic method of controlling the moisture content of the steam in the l.p. cylinders of the main turbine.

The necessity for trouble free gas seals to prevent the escape of toxic gas from the reactor pressure circuit is stressed.

A section of the paper is devoted to the more important aspects of circulator control and the influence of safety consideration upon control philosophy. The self cooling feature of steam turbine driven circulator is outlined and the incorporation and use of pony motors is discussed.

Commissioning tests prior to criticality to prove mechanical soundness of the circulator machinery and freedom from vibration of pressure circuits have an influence upon the type of drive and the purpose of various tests performed at site is described.

A review is included of the circulator and drive features of civil power reactors designed in the U.K. which demonstrates that development and design philosophy has naturally fallen into 3 broad stages.

The first stage involved electric motor drive and the principle of coolant gas flow control by means of speed variation whilst the second stage had variable speed steam turbines or fixed speed electric motors with valve control of the gas flow.

Monolithic concrete pressure vessels containing the reactor core, boilers and circulators currently being constructed constitute the third stage and pose new problems. The diameter and number of vessel penetrations must be minimal. Gas flow control by valving is difficult to achieve for regulating devices would have to be within the vessel with the circulators. Flow control by speed variation has again become attractive. The paper is concluded with a brief discussion on future trends.

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Abstract

EXPERIENCE WITH BRADWELL AND LATINA ON-LOAD FUEL HANDLING EQUIPMENT AND ITS INFLUENCE ON FUTURE DESIGNS

by

*J. O. Joss
+W. H. Shipley
*G. H. Branch
*J. E. Taylor

On-load fuel handling equipment at Bradwell and Latina

1. Description of the Bradwell Machinery

An outline description of the main features of the charge machine and its ancillary plant, especially of those features which required extensive development or modification during the period of design, construction and commissioning before becoming reliable devices.

2. Development of the Design for Bradwell

The original design contained features which it was realised would have to be proved by prototype testing and development. In addition, changes were required because of changes in other reactor plant items. These two factors profoundly influenced the eventual form of certain components.

3. Commissioning at Bradwell

The fuel handling plant test programme was an integral part of overall reactor commissioning with cold, hot and endurance phases, each designed to prove aspects of the detailed and general design. The defects which arose were mainly rectified in situ, but some required further development.

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* The Nuclear Power Group, Knutsford, Cheshire.

+ Messrs. Strachan & Henshaw Ltd. Bristol.

4. Early Bradwell Operational Experience

Because of certain limitations found during commissioning the machinery was at first used off-load or at reduced power and temperature. Under these conditions absorbers and fuel channels were handled without trouble, and considerable experience was obtained in the use of the machinery.

When improved chutes became available, absorber handling at full power was commenced.

5. Full Operational Use of the Bradwell Machinery

After a period of absorber handling it was decided to carry out a controlled approach to full-load fuel handling. The Generating Board produced a programme for refuelling at 20%, 60% and then full power, which was carried out without incident. Mechanical trouble with the fuel grabs caused subsequent refuelling to be suspended while further development of the grabs was undertaken. Meanwhile absorber handling continued, and an incident arose when part of a shield plug assembly broke loose; as a result all such plugs required modification.

Improved grabs, became available, and full-power refuelling was resumed. Mechanical troubles persisted, but an improved indication system prevented these from developing into major incidents. Continued refuelling and further work on grab design continues in parallel.

6. The Latina Machinery as a Development of Bradwell

While basically the same, the Latina charge machines have taken full advantage of the Bradwell design, development, testing and commissioning experiences. As a result of this, much less time was spent in setting the equipment to work on site.

7. On-load Working at Latina

Virtually no on-load handling of fuel elements has yet been carried out, but a considerable number of absorber channels have been changed without incident.

8. The Dungeness and Oldbury Machinery

Following the pattern of continued evolution, the more-or-less identical Dungeness and Oldbury machinery is based on the Latina design, incorporating features to eliminate disadvantages which became apparent through the operation of the earlier designs. Much greater use has been made of test rigs giving improved similation of reactor conditions in the proving in prototype and production form of component parts and complete machines.

9. Lessons for the Future

The paper shows the main requirements for good fuel-handling machinery design, not all of which were obvious when Bradwell and Latina were designed. Much of this knowledge is applicable to AGR as well as to Magnox reactors.

10. Control by Computer

In addition to the mechanical requirements, improvements in control systems can add much to safety, speed and convenience of operation. The application of a digital computer to a refuelling system of the Bradwell type is described and discussed.

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Abstract

THE DESIGN AND CONSTRUCTION OF PRE-STRESSED CONCRETE PRESSURE VESSELS WITH PARTICULAR REFERENCE TO OLDBURY NUCLEAR POWER STATION

by

* A. Houghton Brown
* J.D. Hay
* R.B. Hyde
\+ T.W. Spruce

The paper is introduced by a brief historical review of the development of commercial nuclear power stations in the U.K. which led to the adoption of pre-stressed concrete pressure vessels containing the whole pressure circuit for the coolant gases.

The basic design requirements are stated and expounded coupled with a brief summary of principles for structural behaviour of the vessel in the elastic range and at ultimate failure. The general design for the liner and cooling system is outlined. The requirements to be met by materials and pre-stressing systems are mentioned.

The interrelation of the vessel shape, pre-stressing methods and plant arrangement is examined for spherical and cylindrical vessels, and the costs compared. The choice of end closures and pre-stressing systems for cylinders is discussed and reasons given for the authors' preferences as adopted at Oldbury.

The general theme is then illustrated by reference to the design and construction of the Oldbury station.

A brief description of the vessel includes main dimensions and arrangement of the pre-stressing system. The methods used for elastic and ultimate analyses are outlined and also the design of the liner, the penetrations, the cooling system and the insulation.

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\+ The Nuclear Power Group, Knutsford, Cheshire.

A description is given of the work undertaken in concrete research and in tests of construction feasibility. Further tests included those of components for the stressing system, liner details and models of end slabs and complete vessels.

An outline description is given of the construction of Oldbury which commenced in 1962.

This covers constructional plant, arrangement of foundations and preparation of the base slab of the vessel. This is followed by the roll-in of the prefabricated part of the liner, and the completion of the vessel liner and concreting. The pre-stressing technique is then described.

The proposals for the proof pressure testing are set down.

The paper concludes with a summary of the advantages of pre-stressed concrete pressure vessels for the containment of nuclear reactor circuits, and an appraisal of their potentialities.

The principal factors influencing the choice of design criteria for prestressed concrete pressure vessels are discussed, and permissible stresses in concrete, prestressing steel and bonded reinforcement are considered. The behaviour of the vessel during its life under operating conditions and the deformations imposed upon the vessel and penetration liners are discussed, together with the determination of ultimate load factor.

The principal problems encountered in the design and construction of a prestressed concrete pressure vessel system are described, together with the research which has been undertaken in obtaining design solutions. Individual problems are examined, including vessel stresses and deformations, concrete properties, stress concentrations and penetration reinforcement, bonded reinforcement, prestressing components, behaviour and design of the liner and closures, the cooling system and insulation. The behaviour of model vessels at operating and ultimate load conditions is then described.

The manner in which this research and development has been incorporated in the design of the Wylfa reactor vessels is stated.

A description is given of the twin 96 ft. diameter vessels being built at Wylfa, and of the way in which the main reactor plant is accommodated.

The paper concludes with a brief assessment of the possible future development of the prestressed concrete pressure vessel for nuclear power stations.

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Abstract

THE DEVELOPMENT OF BURST CARTRIDGE DETECTION SYSTEMS WITH PARTICULAR REFERENCE TO SIZEWELL NUCLEAR POWER STATION

by

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Operation of the Calder Hall and Chapel Cross reactors has shown that burst cartridge detection systems must be designed to detect and locate fast and rapid bursts and cavitation failures. The necessity for rapid detection of fast bursts lays emphasis on frequency of scan at the expense of sensitivity. On the other hand, the need to locate channels containing cavitation failures, which contribute to a slow rise in background count, requires the ability to scan the reactor at high sensitivity.

For recent reactors a dual scanning system has been adopted. In this a rapid scan of groups of up to 32 channels provides the routine operational protection and a slow, periodic, scan of individual channels provides a long term record for every channel from which an increasing count can be determined.

In the case of Sizewell the group system is made continuous although the output of the precipitator is presented as a count integrated over a period of about one minute. This reduces the time taken to confirm the existence of a burst and eliminates the need for a selector system, thereby improving reliability of operation.

The reactor refuelling equipment at Sizewell, differs from earlier stations in that charge chutes remain permanently installed in the reactor and guide tubes above the core bring the channel outlets to a configuration suitable for a simple chute. This arrangement enables removable selector valves to be

installed adjacent to the sampling points. These manifold the group samples and operate as selectors for single channel sampling.

The group monitor is provided with high count alarms and only following receipt of such an alarm are the integrator outputs recorded. Data from a single channel scan are recorded on punched cards for subsequent analysis by computer. When the system is being used for burst location, the counts are immediately available to the reactor engineer. Facilities for locking on to particular channels are provided on the basis of 1 facility for every 1000 fuel channels. A bulk gas monitoring system is also provided to back up the other systems, particularly when any channel is not being monitored by a system with alarms.

Advantage has been taken of the existence of a multi-cell precipitation unit developed for the Windscale A.G.R. The performance of the group system was dictated largely by this unit and by the need to keep sample flows small in view of the very large number of samples being taken simultaneously.

The principal items of equipment developed specifically for the Sizewell reactors were the primary and secondary selector valves. The former were required to operate in reactor hot gas conditions with considerable reliability and a minimum of two years between maintenance periods. The secondary valve was developed from the primary valve but is subject to less arduous conditions.

A development of the Sizewell system includes a thermocouple in the primary selector valve enabling the outlet gas temperature of every channel to be measured.

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Abstract

SOME ENGINEERING PROBLEMS OF THE S.G.H.W. 100 MW(E) PROTOTYPE

by

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\+ P. J. Cameron
= N. J. G. Worley

1. The paper discusses some of the more novel engineering features of the S.G.H.W. reactor described by Fawcett, Firth and Holmes(1)

2. The problem of joining the zirconium pressure tubes to the steel primary circuit has been investigated and two types of mechanical joints, i.e. rolled and bolted, have been successfully developed and tested.

3. Dual moderation leads to a tight pressure tube lattice pitch and the engineering solution adopted is described. The results of temperature distribution tests in the bulk moderator are given.

4. Superheat channels are provided to demonstrate that future S.G.H.W. power reactors can exploit the advantages of superheat without sacrificing prototype safety or availability. In these channels, gamma heat generated in the zirconium pressure tube is removed by circulating boiling water through an internal annulus, operating in parallel with the main boiling channels. A thin stainless steel liner prevents convective heat transfer from the superheated steam to the annulus.

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* U.K. Atomic Energy Authority, Reactor Group, Risley
\+ The Nuclear Power Group, Knutsford
= Babcock and Wilcox, London

5. Experimental loop facilities are described, these being provided for one "cluster" fuel element under boiling water conditions associated with a carbon steel circuit, and two "pencil" fuel element channels associated with a stainless steel circuit. Provision is made to discharge coolant from one superheat channel to the condenser when testing an advanced or intentionally defected superheat fuel cluster.

6. Refuelling of the reactor during operation presents a number of design problems requiring novel solutions. Test work leading to solutions for the main problems is outlined.

7. The parameters affecting primary circulating pump run-down time and the means by which the required inertia can be built into the system are outlined. The reasons for choosing a "wet" winding glandless circulating pump are given.

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Abstract

A RESEARCH PROGRAMME TO INVESTIGATE THE SERVICE PERFORMANCE OF STEEL REACTOR VESSELS

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* E. Procter

The paper is concerned with research of an engineering nature in the field of nuclear pressure vessels and primary coolant circuit components. It outlines the problems which are of prime interest to the Central Electricity Generating Board - those concerned with the safety and integrity of the plant throughout its working life. It describes the work so far completed in this direction, and reviews what still needs to be done.

At the present time, it is generally not possible to predict, by calculation, the performance of pressure circuit components in service, especially if creep and other forms of inelastic behaviour are involved. Thus, in the immediate future, the information required must come from tests on actual components and on models, carried out, where necessary, under simulated environmental conditions.

Investigations of this type dealt with in the paper include:-

(a) Three tests beyond yield on highly stressed components - a duct corner unit, a standpipe nozzle cluster, and an intersection between a support skirt and a spherical shell.

(b) A creep test on a standpipe nozzle cluster.

(c) Two cyclic loading tests on a standpipe nozzle cluster and an intersection between a support skirt and a spherical shell.

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Abstract

THE IRRADIATION BEHAVIOUR OF URANIUM AND URANIUM ALLOY FUELS

by

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Basic mechanisms for the two irradiation induced phenomena of growth and swelling have recently been proposed at Harwell. Growth is ascribed to the selective condensation of vacancies and interstitial atoms in preferred crystallographic planes under the anisotropic thermal expansion stresses in the "fission spikes", and it has been deduced that growth effects persist to higher temperatures with increasing fission rate. It has been proposed that swelling due to irradiation-produced inert gases in uranium is governed largely by surface diffusion of the metal atoms, which allows the bubbles to migrate bodily and to collide and coalesce with one another. Theories based on this mechanism can account for breakaway swelling and grain boundary phenomena and do not require any re-solution of the inert gases.

During capsule irradiation tests, pure uranium swelled badly in the temperature range 500-650°C. Small additions of carbon, molybdenum and aluminium reduced the swelling, while iron, chromium, titanium and niobium had no significant effect. The lowest swelling occurred in "adjusted" uranium, i.e. the U.K.A.E.A. Springfields Factory product containing 200-500 ppm iron, 500-1200 ppm aluminium and 300-1200 ppm carbon, beta-quenched and alpha-annealed to achieve small random grains which minimise irradiation growth and surface wrinkling. It contained a fine intermetallic precipitate, primarily $UA1_2$. After comparable irradiations the gas bubbles were much smaller and more numerous than those in pure uranium, and since the pressure in small bubbles is high, the volume increase was less. The fine precipitate is believed to be responsible for the small bubble size and superior swelling resistance, probably by preventing the migration of the bubbles. The thermal cycles to be expected in

a "Magnox" reactor do not affect the swelling resistance of "adjusted" uranium.

Enhanced swelling, due to large grain boundary fissures, was observed in alpha-uranium irradiated at 400-500°C and at fission rates of 15 MW/t or more. This has been called enhanced grain boundary swelling and is believed to be due to irradiation growth stresses probably acting in conjunction with the grain boundary bubbles to produce large tears. "Adjusted" uranium was again found to be superior to pure uranium in resisting this effect, which is not significant at the low fuel ratings used in the "Magnox" reactors as the growth stresses do not persist to sufficiently high temperatures.

Uranium/plutonium/molybdenum alloys irradiated in the gamma-phase at 600°C showed increasing swelling with increasing plutonium or decreasing molybdenum content. However, although the swelling was low compared with pure uranium, the best uranium/plutonium/molybdenum alloy was inferior to "adjusted" uranium.

It is concluded that "adjusted" uranium shows excellent resistance to irradiation growth and swelling which it achieves with only minor additions and without the large nuclear and chemical processing penalties associated with more highly alloyed uranium. It is considered that the irradiation characteristics of this "adjusted" uranium will not limit the life of the present "Magnox" fuel below 5000 MWd/t average.

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Abstract

DEVELOPMENT OF MAGNESIUM ALLOYS FOR FUEL ELEMENTS IN THE UNITED KINGDOM

by

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* R. G. Coffin
= G. C. E. Olds
ß B. Pickles
xx T. C. Wells

For British civil reactors Magnox AL80 has been employed extensively for fuel element canning and structural components. To meet more stringent operating conditions and greater fuel element burn-up requirements, magnesium alloys have been developed with high creep strength and adequate ductility at reactor temperatures whilst having a neutron capture cross section of the same order as that of AL80. In addition, the properties of the magnesium-zirconium alloy ZA have been extensively investigated in order to assess its suitability as a canning material.

The work on ZA has shown that in most respects the properties of this alloy are superior to those of AL80 when considered for canning purposes. Diffusion of plutonium through ZA can walls during reactor service, however, would lead to early discharge of ZA canned elements. Interlayers between can and fuel are being examined in order to prevent such diffusion. Diffusion of plutonium does not occur in AL80 and burn-ups up to 5,470 MWD/Te maximum have been achieved using fine grained AL80 cans.

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* United Power Company, R. and D. Laboratories, Hounslow.
= Associated Electrical Industries Central Research Laboratory, Rugby.
ß United Kingdom Atomic Energy Authority, Culcheth.
xx International Research and Development Company Ltd., Newcastle.

Heat treatment of ZA for 96 hours at 600°C produces a marked increase in the creep strength of the material and renders it suitable for use as splitters and braces in which high creep strengths rather than high ductilities are required. Zirconium containing magnesium alloys, however, are not metallurgically stable in moist carbon dioxide, zirconium hydride being precipitated by reaction with moisture. Such precipitation could lead to changes in properties during service and this has led to the examination of pre-hydrided ZA which is not susceptible to further modification in properties due to hydriding during reactor operation. The properties of pre-hydrided ZA are markedly affected by the stage in manufacture at which the hydriding takes place but it has been shown that WHP ZA (extruded, homogenised for 96 hours at 600°C, hydrided) has a much higher creep strength than ZA heat treated in any other way.

Additions of manganese to ZA produce further improvements in creep strength and both Magzan 2, containing 0.1 - 0.2% manganese and Magzan 4, containing about 0.4% manganese, have been investigated. Both of these alloys contain precipitated phases and Magzan 4 has not yet been produced with satisfactory homogeneity. Most attention has therefore been given in Magzan 2 which gives an excellent combination of high strength and ductility with a fine, stable grain size.

For end caps, in which the volume of material is small and consequently the neutron capture cross-section comparatively unimportant, Mg-Mn alloys containing 0.5 - 1.5% manganese have been investigated. In these alloys particularly, grain size plays a very large part in determining the creep properties, a grain coarsening treatment effecting improvements in creep strengths but lowering ductilities. With the higher manganese alloys, however, good combinations of high strength, ductility and moderate grain size can be achieved.

Preliminary investigations on magnesium alloys manufactured from powder indicate that these can be produced with a very high level of creep strength and adequate ductility in short term tests but much remains to be done both in improvement of production methods and determination of their properties.

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Abstract

THE DEVELOPMENT AND EVALUATION OF URANIUM MAGNOX FUEL ELEMENTS

by

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ø J. Price

1. The paper describes some of the main features of the out-of-pile mechanical development of the magnox-uranium fuel elements in use or to be used in U.K. power reactors, and covers the three year period to December 1963. It is devoted entirely to the development of the stacked design of element, and deals principally with the problems associated with gas forces and stacking loads. Experimental and analytical techniques for the evaluation of fuel element designs have been developed in these fields. They are described and the criteria by which an element is judged acceptable are given. Certain environmental problems, for example thermal cycling fatigue, are not discussed but some idea of the work involved in the development and proving of a fuel element is given.

2. Stability and endurance requirements for the polyzonal fuel element have led to the introduction of a spring stabilising device and a braced splitter assembly with the splitters reduced in height between braces. The local reduction of splitter height reduces the element disturbing forces and virtually eliminates the net splitter force in these regions, and so eases the task of stabilising the element and supporting the splitter. Bowing restraint is provided by the splitter and brace structure at brace positions.

3. The spring is designed to give a force, which, after relaxation caused by irradiation and thermal effects during service, is adequate for complete

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= U.K. Atomic Energy Authority, Reactor Fuel-Element Laboratory
ø English Electric Co. Ltd., Whetstone

stability. It is expected that fin tip movements and splitter tip movements at a brace caused by both gas and bowing loads will be limited to values such that the resulting degradation of heat transfer performance is small and acceptable.

4. During the development period under review the demand for increased heat transfer performance has taken helical can manufacture to the limit of economic feasibility. The chosen design solutions to the problems of gas forces and stacking loads have been consolidated and whilst there is no doubt that alternative solutions are possible, the incentive in terms of performance benefit for developing them is small. On the other hand the herringbone or chevron element appears attractive in that it offers a reduction or elimination of fittings external to the can and should possess an inherent stability. This type of element is being assessed using similar techniques to those developed for the polyzonal element. It has yet to be demonstrated that stability can be achieved without a mechanical device; even so the design has merits which have resulted in its adoption for the second reactor at Sizewell.

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Abstract

POST-IRRADIATION EXAMINATION OF C.E.G.B. FUEL ELEMENTS

by

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*B. F. Sharpe

The aim of our examination is twofold. Firstly, to assist the reactor operator in the elimination of reactor shut-downs; these may be due to perforations in the fuel element can or physical change in the fuel element, either of which may adversely affect the operation of the reactor, or impede discharge from the reactor. Secondly, to provide information on the condition of the heat transfer surfaces of the can together with the extent of deformation of the uranium bar. Upon such data, reactor operation and the future design of fuel elements can be based.

The paper describes the techniques of remote handling, dismantling and examination of fuel elements which have been established to ensure that the maximum amount of information is obtained from a minimum number of remote operations. Attention is drawn to the ability resulting from the use of simple equipment, to change the emphasis of the examination.

Methods of optical examination are outlined and illustrations are given of the way in which deformation of fuel element components and the external surface of the can is assessed. The extent to which radiography can provide useful information is discussed and it is shown that precise measurements of fuel bar distortion can be made from radiographs when the geometry of projection from a point source is taken into account.

The requirements for leak detection in irradiated fuel elements are considered and it is shown that the ideal is most nearly achieved by bubble leak location which involves reducing the pressure above the surface of a liquid in which the element is immersed. Developments of this method permit

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the assessment of leak size and shape.

A description is given of techniques of sectioning fuel elements for metallographic examination, whereby distortion of the specimen is minimised. Facilities for investigating the properties of fuel element materials are described and illustrated by examples of the work currently in progress.

The problems of maintenance, decontamination and waste disposal, as they affect laboratory practice, are briefly discussed. Specific mention is made of a process for converting the swarf from machining operations into inert glass blocks.

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Abstract

PROGRESS TOWARDS A STANDARD DESIGN OF A.G.R. FUEL ELEMENTS FOR POWER REACTORS

By

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J. S. Nairn+
J. D. Thorn=

The advantages to be gained from the standardisation of fuel elements for a large number of reactors include savings in fabrication costs due to unification of manufacturing plant, savings in development costs due to these being shared, and savings in the design and development costs of the reactor plant, due to the further standardisation possible both in components and design treatment. The results of continuing fuel development can be applied to improve the current standard design and are enhanced in value by being directly applicable in new fuel charges for existing, as well as future, reactors of the type. With the A.G.R. concept, such standardisation is possible over a wide range of reactor sizes.

The design and development programme on A.G.R. fuel assemblies is aimed at the formulation of such a standard concept for power reactor fuel elements, taking the original Windscale A.G.R. fuel design as the starting point. The basis for the selection of parameters and structural design features is discussed including a brief outline of a method for estimating the variations of the can temperature from the nominal in particular cluster designs. A synopsis is given of the data selected as most appropriate for the calculation of gaseous fission products within the can under A.G.R. conditions, this influences the chosen reactor conditions.

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+ U.K.A.E.A., Reactor Group, Windscale Works, Seascale, Cumberland.
= U.K.A.E.A., Reactor Group, Risley, Warrington, Lancashire.

Out-of-pile work undertaken to sort out the likely behaviour of fuel elements and assemblies is described. It includes data on the conditions critical for the appearance of longitudinal wrinkles during creep collapse, and on the behaviour of cans under simulated fault conditions. The behaviour of different assembly designs during thermal cycling is discussed. A few notes are included on the effect of dropping assemblies.

The fuel element irradiation programme in W.A.G.R. is described. The correlation of the performance of those already in the reactor with that expected is discussed. A brief description is given of the facilities available for the examination of irradiated elements. The results from the post-irradiation examination of fuel element stringers from W.A.G.R., as well as from certain loop experiments undertaken in NRX under conditions similar to A.G.R., are discussed. The data covers particularly length changes occurring in-pile, fission product gas release and an unusual chemical attack on the inside surfaces of the stainless steel cans.

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Abstract

FUEL ELEMENTS IN THE DOUNREAY FAST REACTOR

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* P. Higginson
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The original target for fuel elements in the Dounreay Fast Reactor was to enable operation at 60 MW and to achieve 1.2% burn-up (heavy atoms). Following modifications to the design of the element this target has been reached.

The metallic fuel used is an alloy of enriched uranium with 20% molybdenum in an annular form. The fuel is niobium canned. The space between the fuel and can is filled with Na and is deliberately vented to the reactor coolant, NaK.

The real fuel irradiation experience was gained from a series of reactor runs at around 30 MW and from the 60 MW runs. The earlier design of fuel element was taken to a maximum of 0.6% burn-up. Progressive cracking of the fuel occurred from 0.3% burn-up onwards. Volume swelling of the fuel, however, was as predicted from M.T.R. results.

The modification to the element design allowed for 8% diametral swelling within the can. These elements achieved 1.2% burn-up and despite fuel cracking remained externally unchanged.

Corrosion of the niobium can due to oxygen in the reactor coolant has been investigated. The results have indicated that the can temperature must be limited to 480°C to avoid corrosion.

* U.K. Atomic Energy Authority, D.E.R.E.

A few elements have exhibited poor bond flow and fission gas agglomeration. This has caused local overheating on the outer surface of some fuel pieces and in a few instances evidence of surface melting was detected. This phenomenon did not affect the canning of the elements.

Continued development based on these results is taking place and higher burn-ups are believed possible.

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Abstract

DEVELOPMENT AND BASIC CHEMISTRY OF A LARGE SCALE MANUFACTURING PROCESS AND PLANT FOR PLUTONIUM FAST REACTOR FUEL

by

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=F. Brown
zW. C. L. Kent

This paper describes the development work now in progress from which future large scale plants for the production of plutonium fast reactor fuel elements in the U.K. will be designed.

The work of selecting the fuel, and of specifying the fuel element design is now proceeding and although at this stage these cannot be finalised, the work will form the starting point for the fuel element process and plant development.

The purpose of the plant required may be divided into three categories:- (a) raw fuel material production, (b) manufacture of fuel into a compacted form, and (c) fabrication of fuel elements. Consideration of the special requirements of criticality, shielding, containment and active maintenance, as applied to a large scale production facility, leads to the conception of continuous operation using safe by shape linked equipment for the bulk fuel,

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sections (a) and (b), and of a production line largely influenced by pin movement considerations, for section (c).

The development for each of these sections is proceeding by way of inactive and uranium based equipment studies in the first instance, leading to initial versions of plant suitable for use with plutonium, and finally to a full demonstration of the plant and containment on a scale giving approximately the output required to support the prototype fast reactor.

The first section of the plant is based on the preparation of mixed Pu/U oxides by continuous coprecipitation from solution with ammonia, using a novel multistage precipitation technique, the chemistry and theory of which are described. The product is continuously filtered, washed and dried and then subjected to calcination, reduction and surface area adjustment in a single stage. The resulting oxide powder possessing the requisite homogeneity, is then converted into a compacted form in a continuous granulation plant, which includes a novel spheroidisation process. This forms the feed for either a pelleted or a vibro-compacted fuel product. Should carbide be required, it is possible to add to this system, equipment for the oxide/carbon reduction process, and thus the fuel production plant is being developed on very flexible lines.

After sintering to high density the fuel is loaded into cans and plant development is concerned largely with the layout of a system for transferring fuel pins of awkward shape along a series of consecutive loading, sealing, welding, etc. operations at a reasonable production rate.

A demonstration plant embodying all these sections is now in the course of erection at Aldermaston.

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Abstract

PRELIMINARY DEVELOPMENT OF DISPERSED FUELS IN THE U.K.A.E.A.

by

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1. This paper discusses the reasons for developing non-metallic dispersed ceramic fuels in the U.K. and their development status. Two systems have been chosen for detailed development: a dispersion of dense uranium dioxide in beryllia, and a dispersion of coated uranium carbide fuel in silicon carbide.

2. Sintering studies on commercially available BeO powders show that it is not possible to obtain consistently a high density product with high strength and low grain size. A route has therefore been developed to produce suitable powders for this purpose. It involves the precipitation of small crystallite size beryllium sulphate and its calcination under defined conditions. This and other powders have been characterised by physical methods; permeability studies have shed some light on the mechanism of sintering in BeO compacts.

3. Fuelled UO_2/BeO specimens are made by extruding a rod of the mixture simultaneously with a clad of pure BeO. After sintering, the exposed ends are sealed by hot pressing a disc of dense BeO on to them. Preliminary irradiations are being carried out on such specimens.

4. The neutron irradiation of dense pure BeO has been studied. Anisotropic expansion of the BeO lattice leads to the storing of strain energy in a polycrystalline material due to grain boundary constraint. At low irradiation

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temperatures and doses below about 5×10^{20} nvt at least, this strain energy can be estimated from X-ray lattice expansion data. At elevated temperatures or higher doses however, crystal growth due to irradiation-induced defect clusters may contribute appreciably to the strain energy.

5. Uranium carbide spheres for dispersion are made by granulating and spherodising a mixture of UO_2 and carbon powders followed by reaction sintering at controlled rates up to 1600°C. Spheres of UC of about 500μ diameter and oxygen contents less than 0.2% have been made on the 10 kg scale. The particles are coated with pyrolytic carbon or silicon carbide in fluidised beds using methane or methyl-trichlorosilane and hydrogen. The effects of varying the reaction conditions on the quality of the coats are discussed.

6. Self-bonded silicon carbide rods are made by extruding a mixture of silicon carbide, carbon and a binder which is finally treated with molten silicon at 1,500°C. Dense rods (92-95% SiC; 8-5% Si) are produced and some of their physical properties are given.

7. Fuelled carbide rods are produced similarly, the fuel spheres being initially mixed with the other ingredients prior to extrusion. A fuel free layer and end-caps may be produced as in the case of BeO/UO_2 rods.

8. The irradiation of fuelled rods is in progress and irradiation results on unfuelled silicon carbide are discussed.

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Abstract

FABRICATION AND IRRADIATION STUDIES OF UO_2-STAINLESS STEEL AND $(U,Pu)O_2$-STAINLESS STEEL CERMETS

by

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J. D. B. Lambert[+] P. G. Mardon[+]
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Studies of cermet fuels, which have been carried out by the U.K.A.E.A. during the past three years, have been concerned primarily with the development of pin and plate fuel elements for possible application in a fast reactor. For this application the oxide should be present in the concentration range 30 to 50 volume %, distributed in such a way that structural weakening of the matrix by fission fragment damage is minimized.

The matrix and cladding materials selected for the bulk of the work were A.I.S.I. Type 316L steels whose compositions were closely controlled to obtain maximum microstructural stability combined with good creep strength and ductility.

Hot swaging techniques were selected for the preparation of pins since these provide uniform compression in a fairly simple manner. Initial efforts were successful in eliminating the "stringering" normally found in cermet fuels but improvements were required as far as matrix densification, ceramic distribution

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and matrix-to-clad bonding were concerned. A technique was successfully developed to remove these defects; spherical granules of oxide were coated with steel powder, consolidated and sintered, and finally densified and bonded at a high temperature in a multi-die swage.

Hot-rolling, by a modification of the "picture-frame" technique normally used to make cermet plates, led to the preparation of dense, bonded plates with fairly uniform ceramic distributions. These plates were subjected to extensive mechanical, thermal and non-destructive tests.

Specimens of pins and plates prepared in the early stages of the fabrication development have been irradiated in thermal fluxes in the PLUTO reactor at Harwell and in the Dounreay Materials Testing Reactor and subjected to post-irradiation examination, while specimens with improved structures are still being irradiated. Encouraging performances have been obtained from pins with better initial structures.

Plate specimens with non-uniform ceramic particle distributions achieved 5% burn-up at 650°C surface temperature without failure but swelling was apparent beyond 7% burn-up.

The results suggest that the optimum irradiation performance may be achieved by careful control of granule composition and density, and of matrix density and permeability.

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Abstract

MONOCARBIDES AS REACTOR FUELS

by

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Basic studies of the preparation of U-C and (UPu)-C by the carbothermic reduction of UO_2 and $(UPu)O_2$ in vacuum have led to the development of alternative processes for the preparation of carbides. One process, which is carried out in an atmosphere of flowing gas such as argon, appears to be particularly suited to continuous operation to give a product which can be fabricated into various fuel forms by a variety of techniques.

It is now possible to produce fuel pellets of high density by sintering carbide powders in argon or in vacuum at temperatures close to 1500°C provided that the powders are sufficiently fine and are prevented from reacting with moisture or oxygen. Accurate control of the composition of the final product can be achieved by careful control of the feed materials, by blending powders of various carbon contents or by hydrogen reduction of the powders or the pellets. In addition, high density rods of carefully controlled composition have been prepared by skull melting and by centrifugal casting and the effects on the final product, of changes in the melting technique and the nature of the feed material for melting, have been studied. These investigations have been supplemented by work on vibratory compaction, gas pressure bonding and electron beam melting

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techniques. The suitability of these processes for the preparation of fuel for particular application is discussed and information on changes in plutonium content of (UPu)-C materials during fabrication is presented.

The potential of carbides as reactor fuels is considered in the light of recent measurements of physical and chemical properties, phase equilibrium determinations and the known irradiation behaviour. Possible future developments leading to improved irradiation behaviour are suggested and some information on the preparation and properties of cemented and bonded carbides, and carbides containing strengthening carbides in solid solution is presented.

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Abstract

THE BEHAVIOUR OF UO_2 AND OF $(U, Pu)O_2$ FUEL MATERIALS UNDER IRRADIATION

by

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J. B. Sayers*
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This paper includes the results of several recent studies of the course of radiation damage in oxide fuel materials and of pre-irradiation characterisation of $(U,Pu)O_2$ solid solutions. Thermodynamic and crystallographic data indicate that single-phase solid solutions are preserved under strongly reducing conditions, and also under mildly oxidising conditions at high temperatures, if the ratio $Pu/(U + Pu) \leq 0.35$. The thermodynamic results are used to discuss the precise composition reached by oxides during preparation and in use and the chemical form of the fission products.

Crystallographic evidence indicates the formation of stable clusters of defects at doses of 5×10^{18} fissions/cm^3 during irradiation at 70°C and at 400°C; most of the damage anneals in several stages between 200 and 700°C. Volatile fission products - I, Te and Cs - move as fast or faster through sintered UO_2 at 1400°C as do rare gas atoms; the introduction of Pu has little intrinsic effect on gas emission. Electron microscopy reveals the migration of pores to grain boundaries during irradiation. The evidence is consistent with the concentration of fission products at grain boundaries, even at low temperatures. Radial temperature gradients have been measured during irradiation; thermal conductivity (K) values do not change appreciably at doses up to 5×10^{19} fissions/cm^3 and mean temperatures of above 500°C, but K values decrease during irradiation at lower temperatures.

Miniature fuel elements of UO_2 have been irradiated to burn-ups of 2 to 5.7% over a range of centre and surface temperatures. Fission gas release varied

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from 0.2 to 68% but no gross dimensional changes were observed. Evidence of some plastic flow was obtained during irradiation to 5.7% burn-up at estimated temperatures lower than 1300°C.

The behaviour of $(Pu_{0.01}\ U_{0.99})O_2$ solid solutions has been compared with that of UO_2 in irradiations at low burn-up and centre temperatures up to 2300°C, and $(Pu_{0.10}\ U_{0.90})O_2$ was tested along with UO_2 to a burn-up of 0.9% at centre temperatures below 1275°C. The introduction of Pu had no deleterious effect in either case; gas release was similar for the UO_2 and $(U,Pu)O_2$ oxides and the $(Pu_{0.1}\ U_{0.9})O_2$ pellets were less cracked than the UO_2 pellets.

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Abstract

FUNDAMENTAL STUDIES OF MATERIALS PROBLEMS IN GAS-COOLED REACTORS

by

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To maintain reliable and economic operation during the lifetime of a civil nuclear power plant sensibly calls for a sound understanding of the effects of a reactor environment on the properties of its component materials. In this paper examples are given of work undertaken by the C.E.G.B. at its Berkeley Nuclear Laboratories to underwrite the British civil nuclear power programme.

The problems studied fall into three groups:-

those capable of restricting the availability and reliability of operation of a gas-cooled uranium-Magnox reactor. The incidence of fast bursts, currently one in 10^4 fuel elements, could be lowered by at least 100 times by reducing the oxidation rate of uranium. It is shown that a UAl_3 layer formed by the diffusion of aluminium into uranium reduces the oxidation rate of uranium in carbon dioxide by some 200 times without significantly affecting neutron economy.

those capable of restricting fuel element lifetime by setting up a systematic failure mode. Quantitative theories are proposed, supported by experimental evidence, which define a) the conditions necessary to limit distortion when metallic uranium fuel elements are used in a temperature gradient such that both α and β phases are present; b) the circumstances which lead to coalescence of fission-gas bubbles with resultant volume changes in the uranium.

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The creep behaviour of magnesium and Magnox above 250°C is rationalised in terms of a constant activation energy, a variable stress index and a grain size index at stresses above 200 p.s.i. and Herring-Nabarro creep below 100 p.s.i.

those capable of affecting the lifetime of the plant or its operating parameters. The effects of thermal and radiolytic oxidation on the chemical reactivity of nuclear graphite towards air in the absence of fast neutrons is discussed, together with the mechanism of carbon deposition by thermal polymerisation of carbon suboxide: the kinetics of polymerisation and the structure and properties of the polymer are defined.

The electron microscopy of graphite has shown that the dislocation behaviour and structure in graphite are those predicted by conventional theory. It is shown that the detailed dislocation arrangement in polycrystalline graphite accounts for many of the observed mechanical properties. Thermal cycling and low temperature creep behaviour of graphite are predicted. The conditions under which graphite could be expected to support large strains are discussed in relation to the irradiation creep phenomenon. In anisotropic materials there is a continuous irradiation creep due to plastic yielding, whose extent is governed by the relaxation properties of the particular structure. It is shown that in isotropic crystal structure there is only a theoretically practicable transient creep phenomenon due to climb of dislocations and clustering of defects.

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Abstract

CORROSION BEHAVIOUR OF STEELS IN CO_2/CO ATMOSPHERES

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The reactions of mild, low alloy and stainless steels with CO_2/CO atmospheres have been studied in support of the development of power reactors cooled by carbon dioxide.

Protective oxide films have been formed on mild and low alloy steels in CO_2 based environments at temperatures up to 550°C, although under some circumstances the films lost their protective character and the reaction rate increased markedly. Film breakdown could be postponed for a considerable time (e.g. 20,000h) and could be either general or localised. It was associated with temperatures > 350°C, high pressures (i.e. 14.6 atm), the presence of water vapour and to a lesser extent carbon monoxide. The behaviour of a given steel was often inconsistent with respect to breakdown; no correlation could be found with steel composition although rimming steels were most susceptible to localised breakdown.

A basic study of the oxidation and carburisation of a 20 Cr/25Ni/Nb stainless steel (fuel element canning material for the Advanced Gas Cooled Reactor at Windscale (WAGR)) has been carried out in pure CO_2, CO and mixtures of these gases at 600-850°C. Thin protective oxide films were formed in all instances.

+ U.K. Atomic Energy Authority, A.E.R.E., Harwell
* U.K. Atomic Energy Authority, Culcheth
≠ English Electric Co., Whetstone
= United Power Co., Hounslow

Carbon transferred from the gas phase during film growth readily passed through the oxide into the steel whilst carbon deposited on the surface of a preformed oxide film diffused only slowly during a subsequent anneal in an inert atmosphere. The penetration of the carbon into the steel was determined and at temperatures $\geq 750^{\circ}C$ a maximum up to 75μ from the oxide/metal interface was found in the carbon concentration-penetration curve.

The oxidation of WAGR fuel element cans has been studied under simulated reactor operation and fault conditions involving a temperature excursion and ingress of steam and air into the coolant circuit. The behaviour was satisfactory for protracted periods in CO_2 up to $750^{\circ}C$, but at higher temperatures loss of fin profile occurred which could adversely affect the heat transfer properties of the cans. The behaviour was not significantly affected by total pressure of additions of CO, H_2O or air to the gas.

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Abstract

THE PROPERTIES OF ZIRCONIUM ALLOYS FOR USE IN WATER COOLED-REACTORS

by

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Recent work in the United Kingdom on the corrosion, hydriding and mechanical properties of zircaloy-2 and zirconium-2½% niobium alloy is reviewed, with particular emphasis on the use of these materials in pressure-tube reactors.

The resistance to oxidation of the Zr-2½% Nb alloy by pressurised water or steam is slightly inferior to that of zircaloy-2 and is more dependent on its metallurgical condition. Exposure to a high flux of fast neutrons increases the rate of oxidation of both alloys at 300-350°C.

In the SGHWR at Winfrith Heath, the external walls of the pressure tubes may be exposed to moist carbon dioxide containing up to 10 per cent air. Whilst zircaloy-2 is adequately resistant to such exposure, Zr-2½% Nb alloy has extremely high rates of oxidation in some heat treatment conditions.

A particular advantage of the Zr-2½% Nb alloy is that it absorbs less hydrogen than zircaloy-2 during corrosion, whatever the prior heat treatment.

Neither of these alloys would be suitable for use in superheated steam at about 500°C. An investigation aimed at understanding the behaviour of zirconium alloys at this temperature is described briefly.

Experiments with small samples and experience in the manufacture of pressure tubes shows that annealed Zr-2½% Nb alloy has strength comparable to that of cold-worked zircaloy-2, and that higher strength may be achieved by cold-working or heat treatment without serious loss of ductility. Pressure tubes have been

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manufactured from zircaloy-2 with cold-work contents up to 30%, and no greater difficulty has been encountered with 20% cold-worked Zr-2½% Nb alloy despite its increased strength.

The effect of hydrogen on mechanical properties is being investigated. Hydrided material has a higher creep rate than the unhydrided alloys under thermal cycling conditions. The presence of hydrogen in excess of 10 p.p.m. also lowers the energy to failure in impact tests. Preliminary results show that irradiation has the same general effect as hydrogen, lowering the impact energy at any given test temperature. The two effects may be additive for hydrogen contents above about 100 p.p.m.

It is concluded that the use of either alloy as fuel cladding or pressure tubes operating at about 300°C would be technically feasible. The higher strength of the Zr-2½% Nb alloy could be utilised by designing thinner pressure tubes, with a saving in neutron absorption. For a fixed temperature, there is then little difference between the predicted hydrogen contents after a service life of 20-30 years of pressure tubes made from the two alloys: the lower rate of hydrogen absorption of the Zr-2½% Nb alloy would be counterbalanced by the thinner cross-section.

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Abstract

FUEL CYCLE COSTS FOR A LARGE NUCLEAR POWER PROGRAMME

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With the start of construction at the Wylfa site, the 5000 MW(e) programme of magnox reactor installation in Britain is approaching completion. A substantial further expansion during the next decade is expected. Prior to the introduction of the Fast Reactor on a commercial scale much of the additional capacity will probably consist of more advanced types of thermal reactor. Development in the U.K. has concentrated on the Advanced Gas-Cooled and the Steam Generating Heavy Water Reactors. A prototype of the first of these has been operating at power at Windscale for more than a year, and construction of an S.G.H.W. prototype is under way at the Authority's Winfrith Establishment. The two parts of this paper discuss different aspects of the fuel cycle costs anticipated for this programme.

In Part I, the fuel cycle costs of Magnox, A.G.R. and S.G.H.W. are discussed and compared, both average values expected for the first ten years of the installation of any one reactor type and the long term trends that are anticipated. Effects of possible variations in uranium ore and enrichment prices are demonstrated, together with the cost reductions that can arise from improved fuel management policies and irradiation limits. Design trends leading

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to lower costs in the long term are identified.

U.K.A.E.A. work on the physics of plutonium utilisation in these systems has shown that it should be possible to introduce plutonium based fuel into both the A.G.R. and S.G.H.W. systems without major changes in reactor design. The economics of the use of plutonium in this way is discussed and the relation between Pu values in these systems and in a fast reactor discussed and illustrated.

Part II discusses the influence on the fuel cycle costs of the scale of the enrichment, fabrication and reprocessing facilities supporting the programme. With an expanding programme, numerous possibilities exist for inter-linking the fuel cycles of the three thermal reactor systems both through the existence of common facilities and through the use of reject uranium from one system as feed for another or for a diffusion plant. To realise these possibilities and the resultant economies, the fuelling operation must be planned as a whole. The effects of changing fuel design and irradiation targets on the demand for production capacity further complicate the position.

Thus, whilst comparisons of fuel cycle costs for the individual reactor systems are necessary when considering e.g. relative rates of installation, cost estimates of the enrichment and processing components of the costs of each system cannot be considered wholly in isolation. These factors are taken into account in discussing the utilisation of the existing production facilities, their future extension, and the cost variations anticipated in meeting the needs of the British Power Programme.

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Abstract

THE DEVELOPMENT OF THE NEW SEPARATION PLANT, WINDSCALE

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The New Separation Plant, Windscale, is a centralised, high throughput reprocessing plant, designed to extract economically plutonium from natural uranium or low enrichment fuels arising from power reactors. This paper describes the evolution of the process and the techniques used in establishing its performance.

The specification of the process and products are given and against these the paper discusses the relative merits of various processes.

Reasons are given for the decision to install a mechanical decanning unit and a 3-cycle, 20% tri-n-butyl phosphate in odourless kerosene extraction system, in which uranium and plutonium are separated in the second cycle. Factors affecting the choice of ion-exchange or solvent extraction processes for the plutonium purification cycles are also described.

The influence of effluent disposal on process selection is emphasised, the implications of solvent degradation briefly discussed and the principal problems and methods of control of nuclear safety in the plant are summarised.

The techniques used in establishing the selected process and equipment in an economical manner are described in the sequence of computation of the flowsheet, selected laboratory studies, pilot plant trials and equipment tests.

Specific chemical problems which are discussed include uranium-plutonium separation, solvent washing and solvent degradation. The procedures used for the development and testing of solvent extraction flowsheets on a

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miniature pilot plant, which closely replicates process conditions on the large-scale plant, are described. The paper outlines the basic principles of the mechanical decanning unit developed and gives the methods used in design and development of the mixer-settler units selected for the process.

It is concluded that the process and plant is capable of flexible and economic operation with considerable potential increase in throughput both for natural uranium and low enriched (e.g. A.G.R.) type fuels.

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Abstract

THE NEW SEPARATION PLANT WINDSCALE - DESIGN OF PLANT AND PLANT CONTROL METHODS

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The first chemical separation plant at Windscale was designed to reprocess low irradiated fuel from the air cooled Windscale reactors and, after modification and additional plant installation, has reprocessed the fuel from the Calder Hall and Chapelcross reactors.

To deal with future arisings from the expanding U.K. Civil Nuclear Power programme a new separation plant has been built at Windscale. This paper describes the design of the new plant, the methods of control and the associated instrumentation system.

The design philosophy which was applied to the new plant is described and an outline is given of the main considerations in siting, layout, activity segregation, nuclear safety, maintenance, ventilation, shielding and containment.

A description of the plant is given for the various operations of fuel cooling, decanning, dissolving, solvent extraction operations, thermal denitration and liquid waste disposal.

The essentials of plant operation are discussed together with the principal methods of plant control used. This is followed by a discussion of instrument requirements and mention is made in particular of new instrument features which have been incorporated in the design.

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Abstract

IRRADIATION BEHAVIOUR OF STEEL AS A STRUCTURAL AND CLADDING MATERIAL

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*P.C.L. Pfeil
*S.B. Wright

Ferritic steel tensile specimens, irradiated at controlled temperatures in the range of 100 to 350°C in a series of reactor spectra, have been tested at room temperature and the effects on the lower yield stresses determined. The magnitudes of the radiation hardening are a function of the neutron dose, irradiation temperature, neutron spectrum and type of steel but are independent of the neutron dose rate. The increases in tensile lower yield stresses ($\Delta\sigma_y$) at 150°C can be given by $\Delta\sigma_y = A(\emptyset t)^{\frac{1}{2}}_{Ni}$ where $(\emptyset t)_{Ni}$ is the fission neutron dose and A is a constant. The hardening is a maximum after irradiation at 100°C.and decreases after irradiation at higher temperatures due to annihilation or redistribution of the point defects.

The effects/

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The effects of neutron spectrum have been analysed in terms of various models of radiation damage and the results used to predict the irradiation embrittlement of the CALDER HALL pressure vessel steel over the life-time of the reactor. After 5 years operation the predicted increase in ductile-brittle transition temperature is about 5°C irrespective of the model used; this is consistent with the results of tests on monitoring specimens exposed adjacent to the pressure vessel and removed after 5 years operation of the reactor. Comparison of the irradiation embrittlement in thick plate and small specimens shows that no significant specimen size effect exists.

The effects of neutron irradiation are to increase the proof and ultimate stresses and to reduce the ductility of austenitic steels in post-irradiation tensile tests at temperatures below about 500°C. These changes in properties can be correlated with the fission (or fast) neutron dose and result from the introduction of point defects into the lattice by fast neutron-atom collisions.

Further reductions in tensile ductility are obtained on testing the irradiated austenitic steels at temperatures above about 650°C. This high temperature loss in ductility is a function of the thermal neutron dose and the magnitude of the effect is independent of the irradiation temperature. The reduction in ductility is probably due to the formation of the precipitates which have been observed on the dislocations and at the grain boundaries in the irradiated specimens. Pre-treatment of austenitic steels, involving precipitating as much as possible of the carbon and nitrogen out of solution in the steel and refining the grain size, reduces the magnitudes of the high temperature loss in ductility in post-irradiation tensile and stress-rupture tests.

The reduction in ductility is not due to irradiation enhanced diffusion but to a thermal neutron induced transmutation effect; preliminary results suggest that the products of the $^{10}B(n,\alpha)^{7}Li$ reaction are either directly or indirectly responsible for the effects observed.